Duke Energy

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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: McGuire Nuclear Station Unit 1 Docket No. 50-369
McGuire Nuclear Station Unit 2 Docket No. 50-370
Changes to Emergency Plan Implementing Procedures

Attached to this letter are a revised Emergency Plan Implementing Procedure (EPIP) Index, a notice to delete three (3) Emergency Plan Implementing Procedures, and a notice of revision to one (1) Emergency Plan Implementing Procedure. These procedure deletions and revision were evaluated pursuant to the requirements of 10 CFR 50.54 (q). These changes do not constitute a reduction in the effectiveness of the emergency plan and the plan continues to meet the requirements of 10 CFR 50.47 (b) and 10 CFR 50 Appendix E. Duke implemented these changes on March 13, 2002. A copy of these changes is also being sent to the NRC Office of Nuclear Material Safety and Safeguards as per 10 CFR 72.44 (f). Revision bars in the procedure index indicate the procedure deletions. The following procedure index changes, deletions, and revision have been implemented:

EPIP Index Page 1	Dated 3/13/2002
EPIP Index Page 2	Dated 3/13/2002
EPIP Index Page 3	Dated 3/13/2002
DELETE the following three procedures:	
OP/0/B/6200/090	Dated 3/13/2002
HP/1/B/1009/015	Dated 3/13/2002
HP/2/B/1009/015	Dated 3/13/2002
REVISION to the following procedure:	
RP/0/A/5700/019 (revision 04)	Dated 3/13/2002

There are no new regulatory commitments in this document. Duke is also supplying two copies of this submittal to the Regional Administrator of Region II. Questions on this document should be directed to Kevin Murray at (704) 875-4672.

Very truly yours,

H. B. Barron

Attachments

A045

xc: (w/attachment)
Mr. Luis Reyes,
Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

(w/attachment)
Mr. Martin J. Virgilio, Director
Office of Nuclear Material Safety and Safeguards
Mail Stop T-8A23
Washington, D.C. 20555-0001

R. E. Martin, USNRC
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

(w/o attachment)
NRC Resident Inspector
McGuire Nuclear Station

E.M. Kuhr (EC05O)

M.T. Cash, Manager NRIA (EC05O)

Electronic Licensing Library (EC05O)

EP File 111

DUKE

McGUIRE NUCLEAR SITE

EMERGENCY PLAN IMPLEMENTING PROCEDURES

APPROVED: Bryan J. Glover
SAFETY ASSURANCE MANAGER

DATE APPROVED 3/14/02

EPIP Index Page 1	Dated 3/13/2002
EPIP Index Page 2	Dated 3/13/2002
EPIP Index Page 3	Dated 3/13/2002

DELETE the following three procedures:

OP/0/B/6200/090	Dated 3/13/2002
HP/1/B/1009/015	Dated 3/13/2002
HP/2/B/1009/015	Dated 3/13/2002

REVISION to the following procedure:

RP/0/A/5700/019, (Revision 04)	Dated 3/13/2002

EMERGENCY PLAN IMPLEMENTING PROCEDURES INDEX

PROCEDURE #	TITLE	REVISION NUMBER
RP/0/A/5700/000	Classification of Emergency	Rev. 008
RP/0/A/5700/001	Notification of Unusual Event	Rev. 016
RP/0/A/5700/002	Alert	Rev. 016
RP/0/A/5700/003	Site Area Emergency	Rev. 016
RP/0/A/5700/004	General Emergency	Rev. 016
RP/0/A/5700/05	Care and Transportation of Contaminated Injured Individual(s) From Site to Offsite Medical Facility	DELETE
RP/0/A/5700/006	Natural Disasters	Rev. 009
RP/0/A/5700/007	Earthquake	Rev. 007
RP/0/A/5700/008	Release of Toxic or Flammable Gases	Rev. 004
RP/0/A/5700/009	Collisions/Explosions	Rev. 001
RP/0/A/5700/010	NRC Immediate Notification Requirements	Rev. 013
RP/0/A/5700/011	Conducting a Site Assembly, Site Evacuation or Containment Evacuation	Rev. 005
RP/0/A/5700/012	Activation of the Technical Support Center (TSC)	Rev. 019
RP/0/A/5700/013	Activation of the Emergency Operations Facility (EOF)	DELETE
RP/0/A/5700/14	Emergency Telephone Directory	DELETE
RP/0/A/5700/015	Notifications to the State and Counties from the EOF	DELETE
RP/0/A/5700/16	EOF Commodities and Facilities Procedure	DELETE
RP/0/A/5700/17	Emergency Data Transmittal System Access	DELETE
RP/0/A/5700/018	Notifications to the State and Counties from the TSC	Rev. 010
RP/0/A/5700/019	Core Damage Assessment	Rev. 004
RP/0/A/5700/020	Activation of the Operations Support Center (OSC)	Rev. 011
RP/0/A/5700/21	EOF Access Control	DELETE
RP/0/A/5700/022	Spill Response Procedure	Rev. 009
RP/0/A/5700/024	Recovery and Reentry Procedure	Rev. 002
RP/0/A/5700/026	Operations/Engineering Technical Evaluations in the Technical Support Center (TSC)	Rev. 002
RP/0/B/5700/023	Community Relations Emergency Response Plan	Rev. 002
OP/0/B/6200/090	PALSS Operation for Accident Sampling	DELETED

EMERGENCY PLAN IMPLEMENTING PROCEDURES INDEX

PROCEDURE #	TITLE	REVISION NUMBER
HP/0/B/1009/002	Alternative Method for Determining Dose Rate Within the Reactor Building	Rev. 002
HP/0/B/1009/003	Recovery Plan	Rev. 003
HP/0/B/1009/05	Initial Evaluation of Protective Action Guides Due to Abnormal Plant Conditions	DELETED
HP/0/B/1009/006	Procedure for Quantifying High Level Radioactivity Releases During Accident Conditions	Rev. 005
HP/0/B/1009/010	Releases of Radioactive Effluents Exceeding Selected Licensee Commitments	Rev. 006
HP/1/B/1009/015	Unit 1 Nuclear Post-Accident Containment Air Sampling System Operating Procedure	DELETED
HP/2/B/1009/015	Unit 2 Nuclear Post-Accident Containment Air Sampling System Operating Procedure	DELETED
HP/0/B/1009/016	Distribution of Potassium Iodide Tablets in the Event of a Radioiodine Release	Rev. 002
HP/0/B/1009/020	Manual Procedure for Offsite Dose Projections	DELETED
HP/0/B/1009/021	Estimating Food Chain Doses Under Post-Accident Conditions	Rev. 001
HP/0/B/1009/022	Accident and Emergency Response	Rev. 003
HP/0/B/1009/023	Environmental Monitoring for Emergency Conditions	Rev. 004
HP/0/B/1009/024	Personnel Monitoring for Emergency Conditions	Rev. 001
HP/0/B/1009/029	Initial Response On-Shift Dose Assessment	Rev. 005
SH/0/B/2005/001	Emergency Response Offsite Dose Projections	Rev. 001
SH/0/B/2005/002	Protocol for the Field Monitoring Coordinator During Emergency Conditions	Rev. 002
SR/0/B/2000/01	Standard Procedure for Public Affairs Response to the Emergency Operations Facility	Rev. 003
SR/0/B/2000/002	Standard Procedure for EOF Commodities and Facilities	Rev. 002
SR/0/B/2000/003	Activation of the Emergency Operations Facility	Rev. 008
SR/0/B/2000/004	Notification to States and Counties from the Emergency Operations Facility	Rev. 004

EMERGENCY PLAN IMPLEMENTING PROCEDURES INDEX

PROCEDURE #	TITLE		REVISION NUMBER
McGuire Site Directive 280	Site Assembly/Accountability and Evacuation/Containment Evacuation		DELETED
EP Group Manual	Section 1.1	Emergency Organization	Rev. 017
MNS RP Manual:	Section 18.1	Accident and Emergency Response	DELETED
	Section 18.2	Environmental Monitoring for Emergency Conditions	DELETED
	Section 18.3	Personnel Monitoring for Emergency Conditions	DELETED
	Section 18.4	Planned Emergency Exposure	DELETED
PT/0/A/4600/088	Functional Check of Emergency Vehicle and Equipment		Rev. 007

(R04-01)

Duke Power Company (1) ID No. RP/0/A/5700/019
PROCEDURE PROCESS RECORD Revision No. 004

PREPARATION

(2) Station MCGUIRE NUCLEAR STATION

(3) Procedure Title Core Damage Assessment

(4) Prepared By Thomas M. Kuncewicki Date 03/06/02

(5) Requires NSD 228 Applicability Determination?

- ☐ Yes (New procedure or revision with major changes)
- ☐ No (Revision with minor changes)
- ☐ No (To incorporate previously approved changes)

(6) Reviewed By _____ (QR) Date 3/07/02

Cross-Disciplinary Review By _____ (QR) NA Date 3/07/02

Reactivity Mgmt. Review By _____ (QR) NA Date 3/07/02

Mgmt. Involvement Review By _____ (Ops Supt.) NA Date 3/07/02

(7) Additional Reviews

Reviewed By _____ Date 3/12/02

Reviewed By _____ Date _____

(8) Temporary Approval (*if necessary*)

By _____ (OSM/QR) Date _____

By _____ (QR) Date _____

(9) Approved By SC Ballard Date 3-13-02

PERFORMANCE (*Compare with Control Copy every 14 calendar days while work is being performed.*)

(10) Compared with Control Copy _____ Date _____

Compared with Control Copy _____ Date _____

Compared with Control Copy _____ Date _____

(11) Date(s) Performed _____

Work Order Number (WO#) _____

COMPLETION

(12) Procedure Completion Verification

- ☐ Yes ☐ NA Check lists and/or blanks initialed, signed, dated, or filled in NA, as appropriate?
- ☐ Yes ☐ NA Required enclosures attached?
- ☐ Yes ☐ NA Data sheets attached, completed, dated, and signed?
- ☐ Yes ☐ NA Charts, graphs, etc. attached dated, identified, and marked?
- ☐ Yes ☐ NA Procedure requirements met?

Verified By _____ Date _____

(13) Procedure Completion Approved _____ Date _____

(14) Remarks (*Attach additional pages, if necessary*)

Duke Power Company McGuire Nuclear Station **Core Damage Assessment** **Multiple Use**	Procedure No. RP/**0**/A/5700/019 Revision No. 004 Electronic Reference No. MC0048MM

Core Damage Assessment

1. Symptoms

NOTE: This procedure will normally be performed by Nuclear Engineers while in the Technical Support Center (TSC) to provide a means of determining the status of the core based on various parameters.

1.1 1(2) EMF 51, "Containment Radiation Monitor" in alarm.

1.2 High Core Exit Thermocouple (CET) readings.

1.3 Low Reactor Vessel Level Indication System (RVLIS) levels.

1.4 Any condition in which failed fuel is suspected.

2. Immediate Actions

None

3. Subsequent Actions

NOTE:
1. Perform each of the following sections as data is available.
2. Test Coordinator may elect to perform some or all of the following sections in the order that best suits the situation.
3. Since the plant status may change during the performance of this procedure it may be desirable to repeat certain parts of this procedure.
4. Results of the various instrument readings are estimates. The results from different methods may disagree.
5. Some of the key OAC points are listed in Enclosure 4.5.

3.1 Initial Estimate of Core Conditions

3.1.1 Figure 1 and Figure 2 (and associated tables).

Figure 1
Containment Radiation Level vs. Time for RCS Release

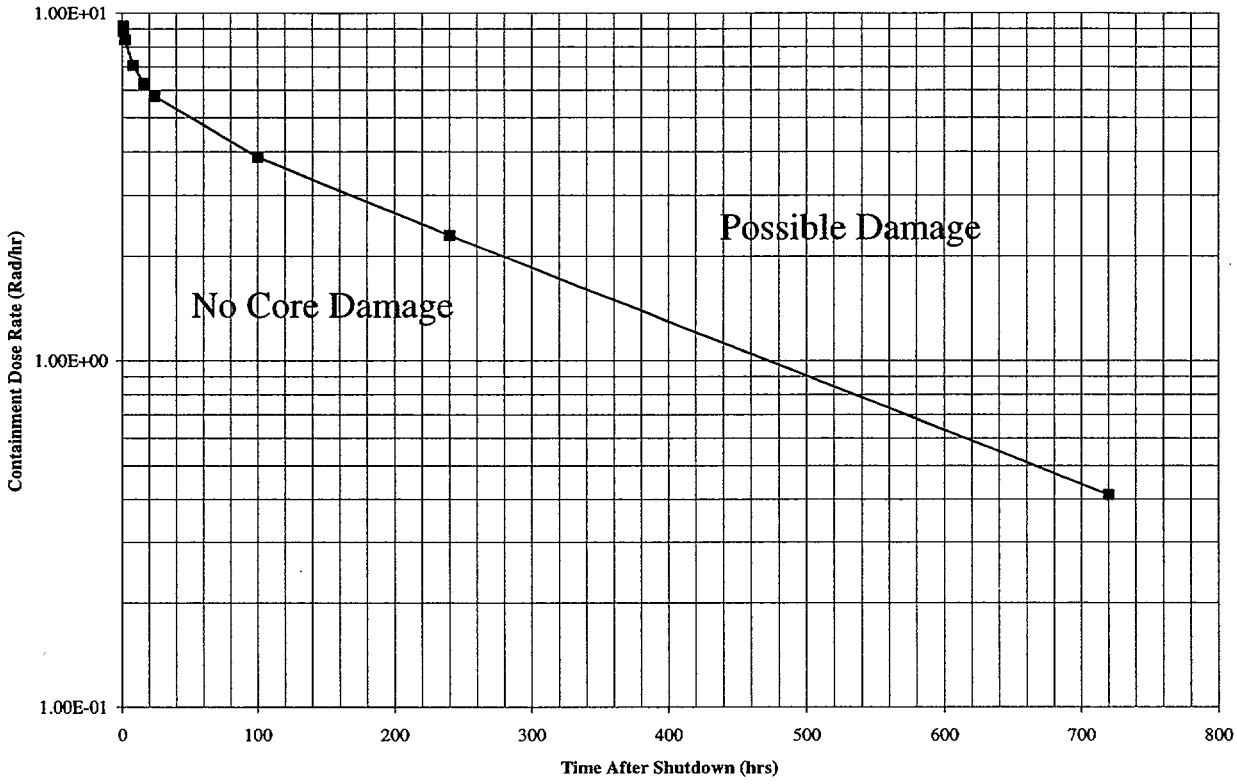


Figure 1	Containment Radiation Level vs. Time for RCS Release
Time After Shutdown (hrs)	Containment Dose Rate (Rad/hr)
0.5	9.1808
1	8.8621
2	8.3792
8	7.0574
16	6.2611
24	5.7672
100	3.8545
240	2.3002
720	0.41169

Figure 2
Containment Radiation Level vs. Time for 1% Fuel Overtemperature Release

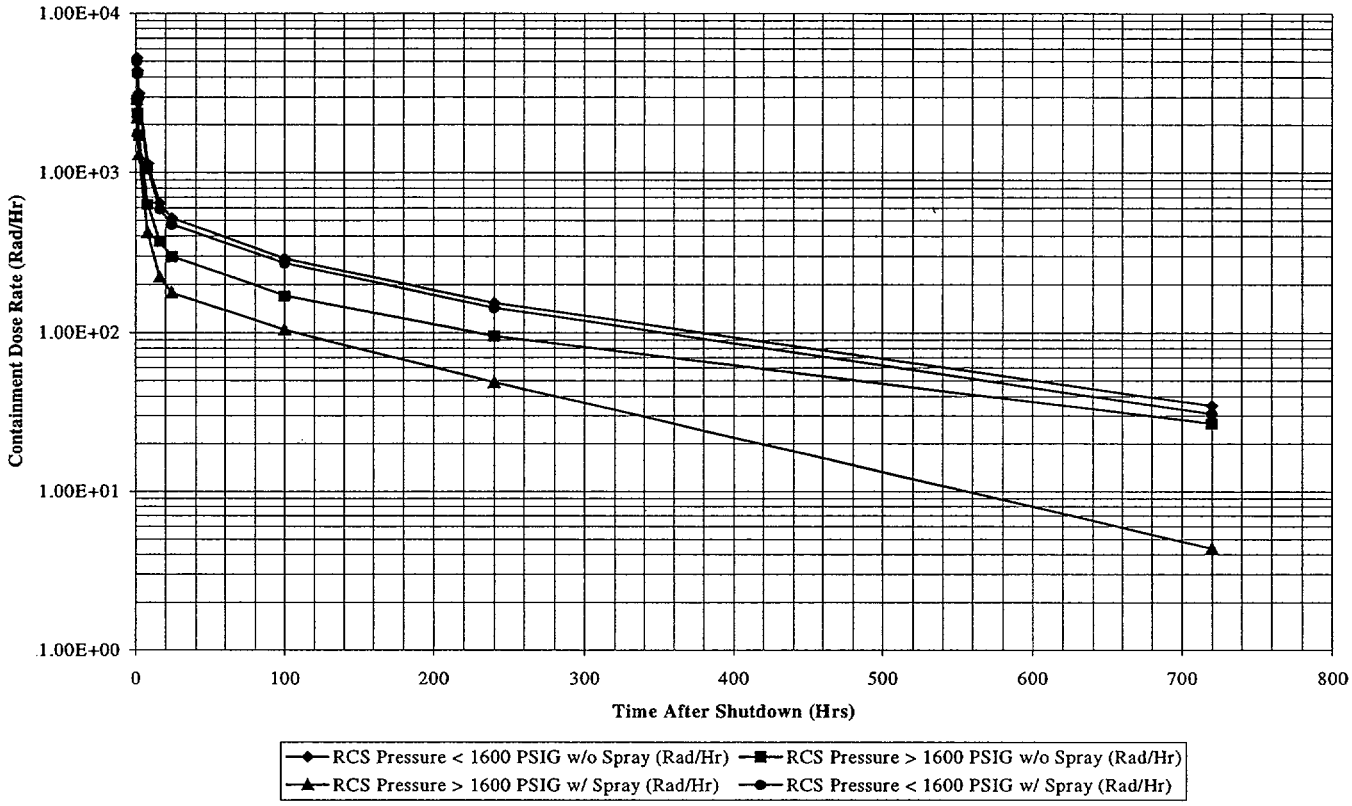


Figure 2	Containment Radiation Level vs. Time for 1% Fuel Overtemperature Release			
Time After Shutdown (hrs)	RCS Pressure < 1600 PSIG w/o Spray (Rad/Hr)	RCS Pressure > 1600 PSIG w/o Spray (Rad/Hr)	RCS Pressure > 1600 PSIG w/ Spray (Rad/Hr)	RCS Pressure < 1600 PSIG w/ Spray (Rad/Hr)
0.5	5.30E+03	2.86E+03	2.19E+03	5.03E+03
1	4.39E+03	2.37E+03	1.82E+03	4.17E+03
2	3.16E+03	1.71E+03	1.29E+03	2.99E+03
8	1.13E+03	6.33E+02	4.20E+02	1.05E+03
16	6.45E+02	3.71E+02	2.21E+02	5.88E+02
24	5.15E+02	2.98E+02	1.78E+02	4.72E+02
100	2.90E+02	1.70E+02	1.04E+02	2.73E+02
240	1.54E+02	9.56E+01	4.88E+01	1.43E+02
720	3.49E+01	2.68E+01	4.37E+00	3.08E+01

_____ 3.1.2 Complete the following table based on Current Plant Data (OAC PID's can be found on Enclosure 4.5).

Time of data, (mm/dd/yy hh:mm)		
Time of Reactor Shutdown, T_0		Hours
Core Exit Thermocouples (CET)		Deg F
RVLIS Lower Range		%
Containment Radiation Monitors 1(2)EMF 51A or B OAC PID A0829 or A0835		R/hr

_____ 3.1.3 Use the above data in 3.1.2 the figures in 3.1.1 to determine the condition that best represents the core.

	No Core Damage	**Cladding Damage**	**Fuel Overtemp Damage**
CET °F	< 700	< 2000	>1400
RVLIS (%)	> 55	< 55 to 40	< 40
EMF (R/hr)	**< Figure 1**	**< Figure 2**	**> Figure 2**

Condition Best Representing Core	Date/Time

_____ 3.1.4 **IF** 'NO CORE DAMAGE' is found, exit this procedure and continue to monitor plant conditions. If conditions warrant, re-run this procedure at any time.

IF 'CLADDING DAMAGE' is found, proceed to Step 3.2.

IF 'FUEL OVERTEMPERATURE DAMAGE' is found, proceed to Step 3.3.

3.2 Clad Damage Assessment

NOTE: EMF 51 may not be useful to assess core damage for containment bypass sequences (e.g., S/G tube ruptures).

_____ 3.2.1 Record 1(2)EMF 51 reading:

_____ R/hr at _____ hours after shutdown.

Containment Spray: ☐ ON ☐ OFF

RCS Pressure: _____ psig

3.2.2 Figure 3 (and associated table)

Figure 3
Containment Radiation Level vs. Time for 100% Clad Damage Release

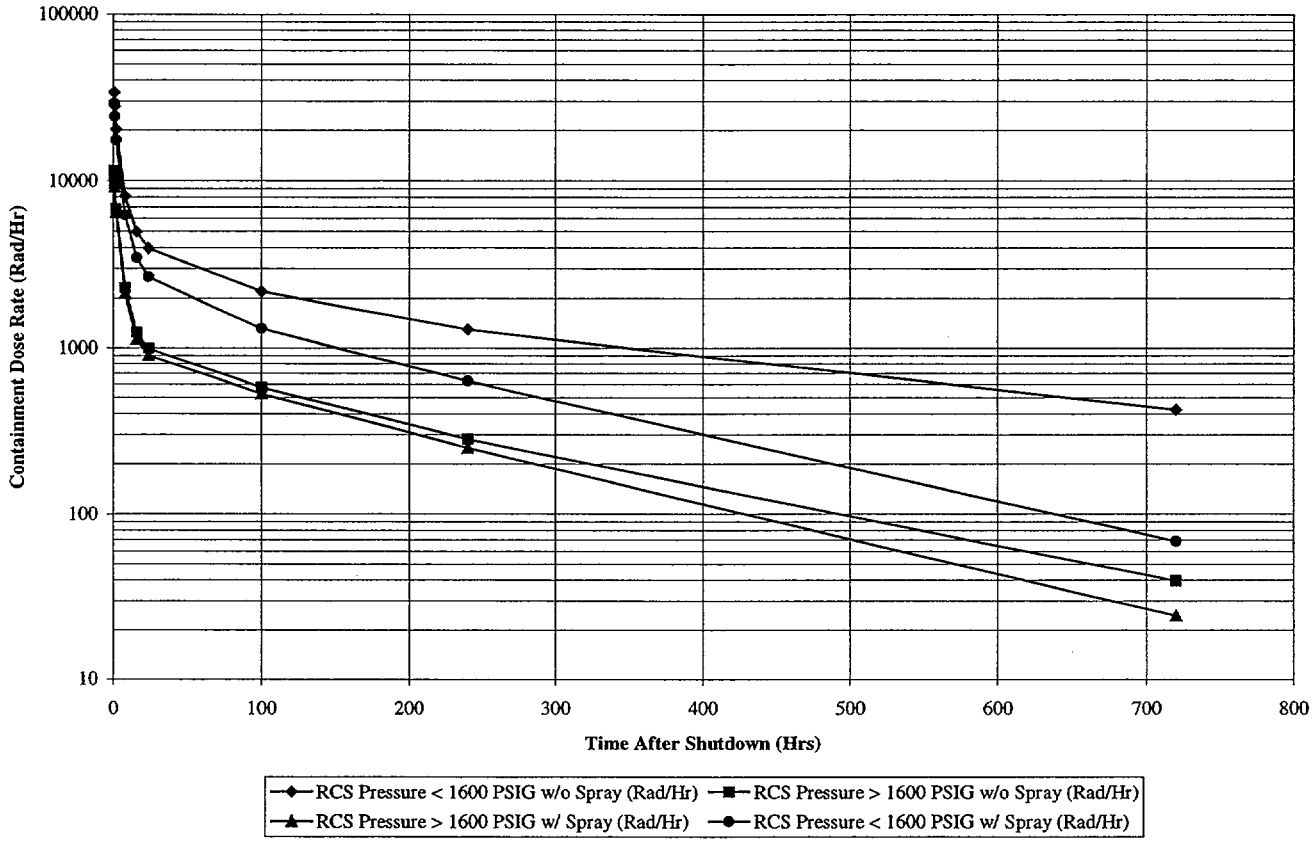


Figure 3	Containment Radiation Level vs. Time for 100% Clad Damage Release			
Time After Shutdown (hrs)	RCS Pressure < 1600 PSIG w/o Spray (Rad/Hr)	RCS Pressure > 1600 PSIG w/o Spray (Rad/Hr)	RCS Pressure > 1600 PSIG w/ Spray (Rad/Hr)	RCS Pressure < 1600 PSIG w/ Spray (Rad/Hr)
0.5	3.37E+04	1.15E+04	1.10E+04	2.91E+04
1	2.78E+04	9.53E+03	9.14E+03	2.42E+04
2	2.03E+04	6.80E+03	6.51E+03	1.75E+04
8	8.09E+03	2.28E+03	2.13E+03	6.26E+03
16	4.96E+03	1.23E+03	1.12E+03	3.48E+03
24	3.98E+03	9.90E+02	9.02E+02	2.68E+03
100	2.19E+03	5.75E+02	5.28E+02	1.30E+03
240	1.29E+03	2.81E+02	2.49E+02	6.28E+02
720	4.22E+02	3.96E+01	2.43E+01	6.88E+01

_____ 3.2.3 Using RCS Pressure from Step 3.2.1 and **Figure 3** (and associated table), find the "Predicted Containment Radiation Level at 100% Clad Damage"

Predicted Containment Radiation Level at 100% Clad Damage: _____

$$\%\ \text{Clad Damage} = \frac{\text{Current Containment Radiation Level}}{\text{Predicted Containment Radiation Level for 100\% Clad Damage}}$$

_____ 3.2.4

$$\%\ \text{Clad Damage} = \text{——————————} \times 100 = \text{_____}\ \%$$

3.2.5 Estimate Fuel Clad Damage based on Core Exit Thermocouple Readings.

_____ A. Complete Enclosure 4.1.

_____ B. **RCS** Pressure **GREATER THAN 1600** PSIG (If Applicable):

Number of CET's **> 1400°F** from Enclosure 4.1: _____ CET's

Number of Operable CET's from Enclosure 4.1: _____ CET's

$$\%\ \text{Clad Damage} = [\frac{\text{Number of CET's} > 1400°\text{F}}{\text{Total Operable CET's}}] \times 100$$

$$\%\ \text{Clad Damage} = \text{——————} \times 100 = \text{_____}\ \%$$

_____ C. **RCS** Pressure **LESS THAN 1600** PSIG (If Applicable):

Number of CET's **> 1200°F** from Enclosure 4.1: _____ CET's

Number of Operable CET's from Enclosure 4.1: _____ CET's

$$\%\ \text{Clad Damage} = [\frac{\text{Number of CET's} > 1200°\text{F}}{\text{Total Operable CET's}}] \times 100$$

$$\%\ \text{Clad Damage} = \text{——————} \times 100 = \text{_____}\ \%$$

_____ 3.2.6 Record Results From Steps 3.2.4 and 3.2.5 onto Enclosure 4.4.

_____ 3.2.7 Confirm Reasonableness of Clad Damage Estimates using Expected Responses below. Record any Comments on Enclosure 4.4.

A. Expected Responses

- RVLIS Less Than **53.5%** and Greater Than **39%**
- Hot Leg RTD Greater Than T_{sat} and Less Than **650°F**
- Source Range Monitor Greater Than **1e+5** cps
- Difference in clad damage estimates from Containment Radiation Monitors (CRM) and CET's less than 50%, using:

$$ABS\left[\frac{\%CladDamage_{CRM} - \%CladDamage_{CET}}{\%CladDamage_{CRM}}\right]$$

B. If the expected response is not obtained, determine if the deviation can be explained from the accident progression

•Injection of water to the RCS

•Bleed Paths from the RCS

•Direct radiation to the containment radiation monitors

or

If the expected response is not obtained, determine if the deviation can be explained from conservatism in the predictive model

•fuel burnup

•fission product retention in the RCS

•fission product removal from containment

_____ 3.2.8 **IF** time permits, complete Enclosure 4.2, Source Range Detector Response.

_____ 3.2.9 **IF** time permits, complete Enclosure 4.3, Movable Detector Surveillance in an Accident Condition

3.3 Fuel Overtemperature Damage Assessment

_____ 3.3.1 Record 1(2)EMF 51 reading:

_____ R/hr at _____ hours after shutdown.

Containment Spray: ☐ ON ☐ OFF

RCS Pressure: _____ psig

3.3.2 Figure 4 (and associated table)

Figure 4
Containment Radiation Level vs. Time for 100% Fuel Overtemperature Release

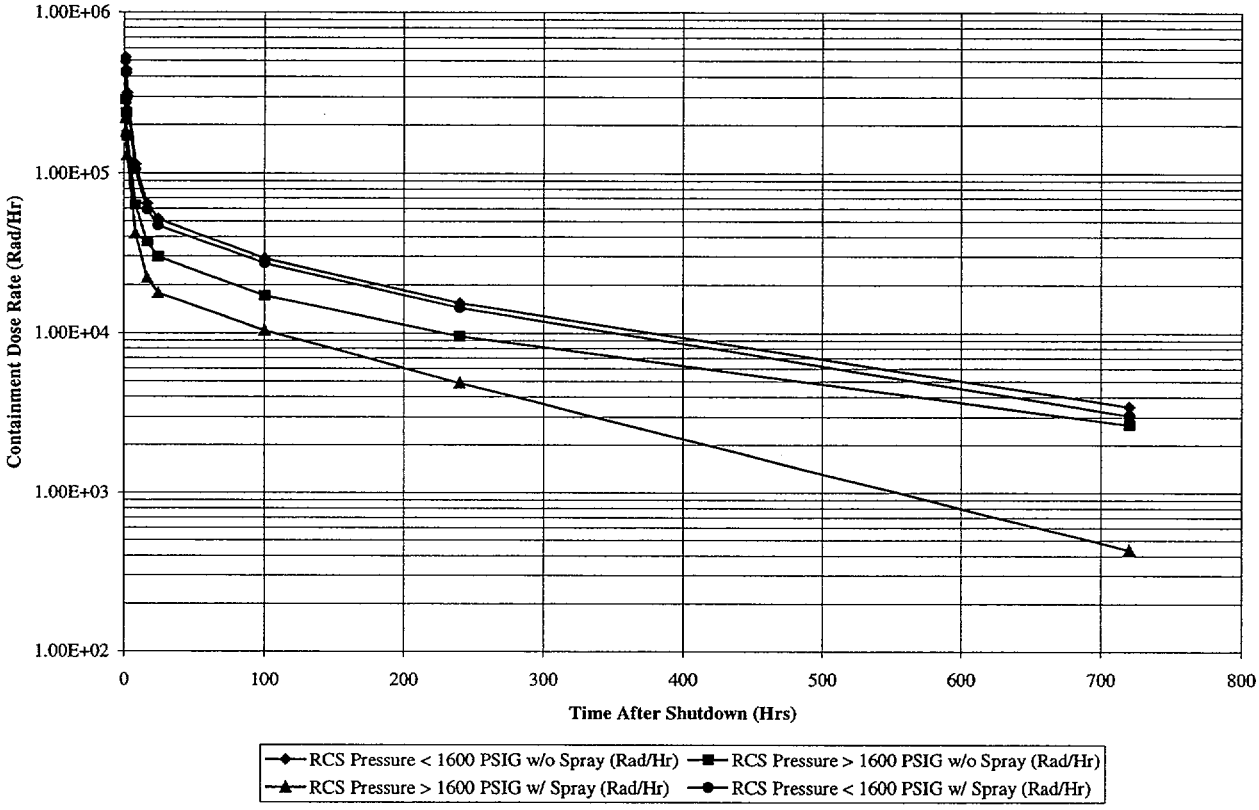


Figure 4	Containment Radiation Level vs. Time for 100% Fuel Overtemperature Release			
Time After Shutdown (hrs)	RCS Pressure < 1600 PSIG w/o Spray (Rad/Hr)	RCS Pressure > 1600 PSIG w/o Spray (Rad/Hr)	RCS Pressure > 1600 PSIG w/ Spray (Rad/Hr)	RCS Pressure < 1600 PSIG w/ Spray (Rad/Hr)
0.5	5.30E+05	2.86E+05	2.19E+05	5.03E+05
1	4.39E+05	2.37E+05	1.82E+05	4.17E+05
2	3.16E+05	1.71E+05	1.29E+05	2.99E+05
8	1.13E+05	6.33E+04	4.20E+04	1.05E+05
16	6.45E+04	3.71E+04	2.21E+04	5.88E+04
24	5.15E+04	2.98E+04	1.78E+04	4.72E+04
100	2.90E+04	1.70E+04	1.04E+04	2.73E+04
240	1.54E+04	9.56E+03	4.88E+03	1.43E+04
720	3.49E+03	2.68E+03	4.37E+02	3.08E+03

_____ 3.3.3 Using RCS Pressure from Step 3.3.1 and **Figure 4** (and associated table), find the "Predicted Containment Radiation Level vs. Time for 100% Fuel Overtemperature Release"

Predicted Containment Radiation
Level For 100% Fuel OT Release: _____

$$\%\ \text{Core Damage} = \frac{\text{Current Containment Radiation Level}}{\text{Predicted Containment Radiation Level for 100\% OT Release}}$$

_____ 3.3.4

$$\%\ \text{Core Damage} = \frac{\qquad\qquad}{\qquad\qquad} \times 100 = \_\_\_\_\_\ \%$$

3.3.5 Estimate Fuel OT Damage based on Core Exit Thermocouple Readings.

_____ A. Complete Enclosure 4.1.

_____ B. Number of CET's **> 2000°F** from Enclosure 4.1: _____ CET's

Number of Operable CET's from Enclosure 4.1: _____ CET's

$$\%\ \text{Core Damage} = \left[\frac{\text{Number of CET's} > 2000°\text{F}}{\text{Total Operable CET's}}\right] \times 100$$

$$\%\ \text{Core Damage} = \frac{\qquad\qquad}{\qquad\qquad} \times 100 = \_\_\_\_\_\ \%$$

_____ 3.3.6 Record Results From Steps 3.3.4 and 3.3.5 onto Enclosure 4.4.

_____ 3.3.7 Confirm Reasonableness of Expected Core OT Damage Estimate using Predicted Responses below. Record any Comments on Enclosure 4.4.

A. Expected Responses for Fuel Overtemperature Damage

- RVLIS Less Than **39%**
- Hot Leg RTD Greater than **650°F**
- Source Range Monitor Greater Than **1e+5** cps
- Difference fuel overtemperature estimates from Containment Radiation Monitors (CRM) and CET's less than 50%, using:

$$ABS\left[\frac{\%\ \text{Core Damage}_{CRM} - \%\ \text{Core Damage}_{CET}}{\%\ \text{Core Damage}_{CRM}}\right]$$

B. If the expected response is not obtained, determine if the deviation can be explained from the accident progression

•Injection of water to the RCS

•Bleed Paths from the RCS

•Direct radiation to the containment radiation monitors

or

If the expected response is not obtained, determine if the deviation can be explained from conservatism in the predictive model

•fuel burnup

•fission product retention in the RCS

•fission product removal from containment

_____ 3.3.8 **IF** time permits, complete Enclosure 4.2, Source Range Detector Response.

_____ 3.3.9 **IF** time permits, complete Enclosure 4.3, Movable Detector Surveillance in an Accident Condition.

4. Enclosures

4.1 Core Exit Thermocouple Analysis

4.2 Source Range Detector Response

4.3 Movable Incore Detector Response

4.4 Results and Comments

4.5 OAC Points

5. References

5.1 Duke Power Calculation DPC-1507.03-00-0001, "Core Damage Assessment Guidelines.", Rev 0, 6/30/92.

5.2 "Accident Response of Instrumentation" by Westinghouse.

NOTE:
1. Safety grade thermocouples do not need any temperature correction but non-safety grade thermocouples need temperature correction since they are still tied into the reference junction.
2. OAC Program Nuclear, Incore Thermocouple Map, will show both safety and non-safety grade thermocouples.
3. Thermocouple calibration may be incorrect if they experience temperatures in excess of 1650 Deg. F.
4. The calibrated range of the CET system is 32 Deg F to 2300 Deg F. Once core uncovery occurs, CET readings will be considerably lower than the average maximum fuel temperature.

_____ 4.1.1 **IF** possible, obtain a printout of OAC Program Nuclear, Incore Thermocouple Map and attach to this enclosure.

_____ 4.3.2 **IF** desired, use the manual toggle switch panel for CET (in the back of the Control Room on the Moveable Incore Detector panel) to record the non-safety grade thermocouple readings on Page 3 of this enclosure.

NOTE: Each reference junction box, located in containment, has 2 RTDs which feed into the OAC. Either RTD can be used for step 4.3.2.

_____ 4.3.2 Obtain the reference junction box temperature from the OAC (A1265 or A1271) and record below.

A. Reference Junction Box Temperature = _____Deg F

B. Verify Reference Junction Box Temperature is reading 160 ± 100 Deg F. If not, record the difference between the actual junction box reading and 160 Deg F at the Top of Page 3.

Actual Reading – 160°F = _____°F

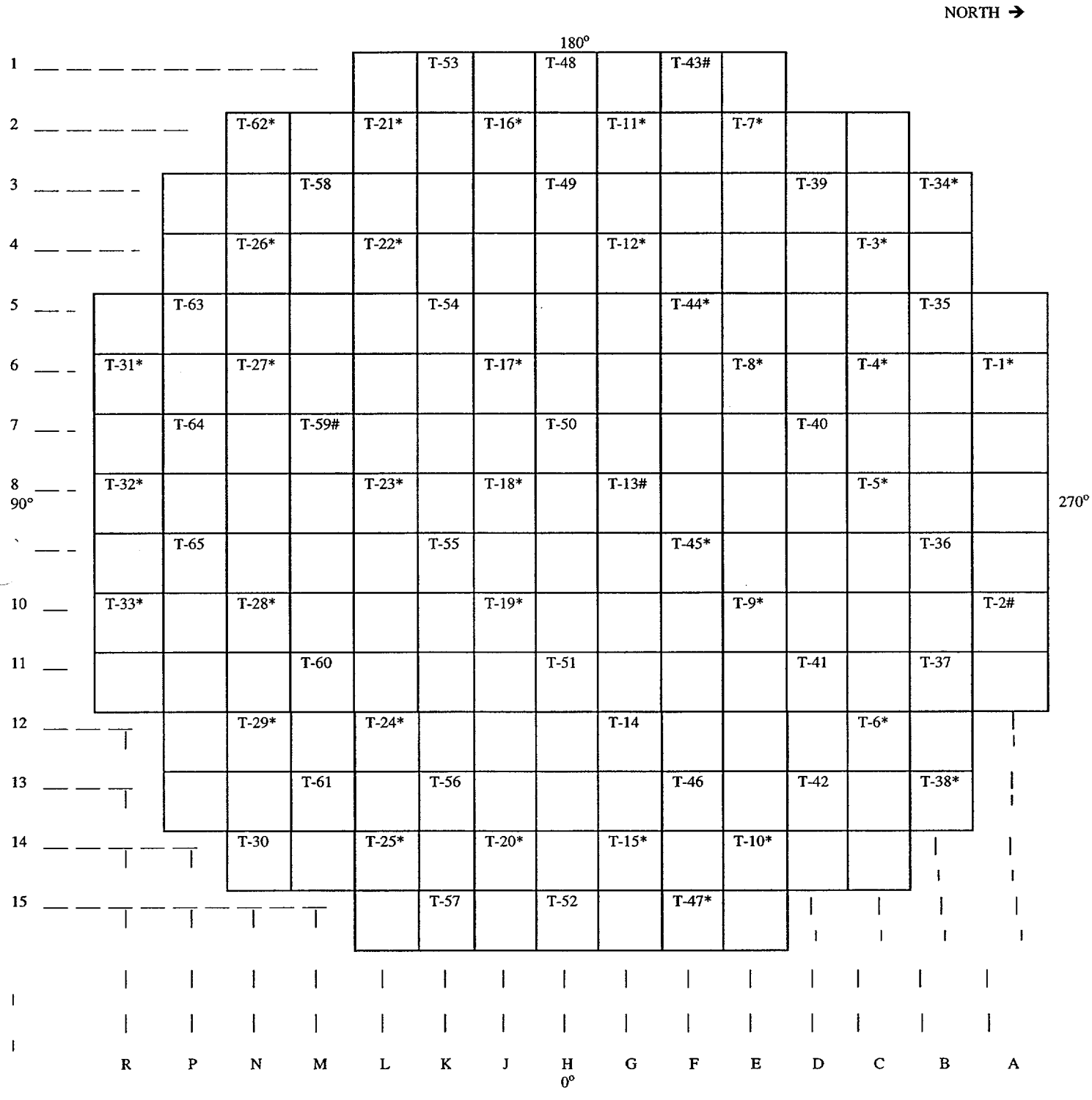
_____°F – 160°F = _____°F

_____ 4.3.3 Complete the **CET Readings** table on Page 3 of this enclosure to record the readings from available CETs.

Enclosure 4.1

Core Exit Thermocouple Analysis

CET Core Locations



T-XX

XX - Thermocouple #
T-XX* Safety Grade
T-XX# These thermocouples are temporarily located in the upper head on U-1 only.

CET Readings

Non-Safety Grade Thermocouples Reference Junction Difference from 160°F (Step 4.3.2) =_____

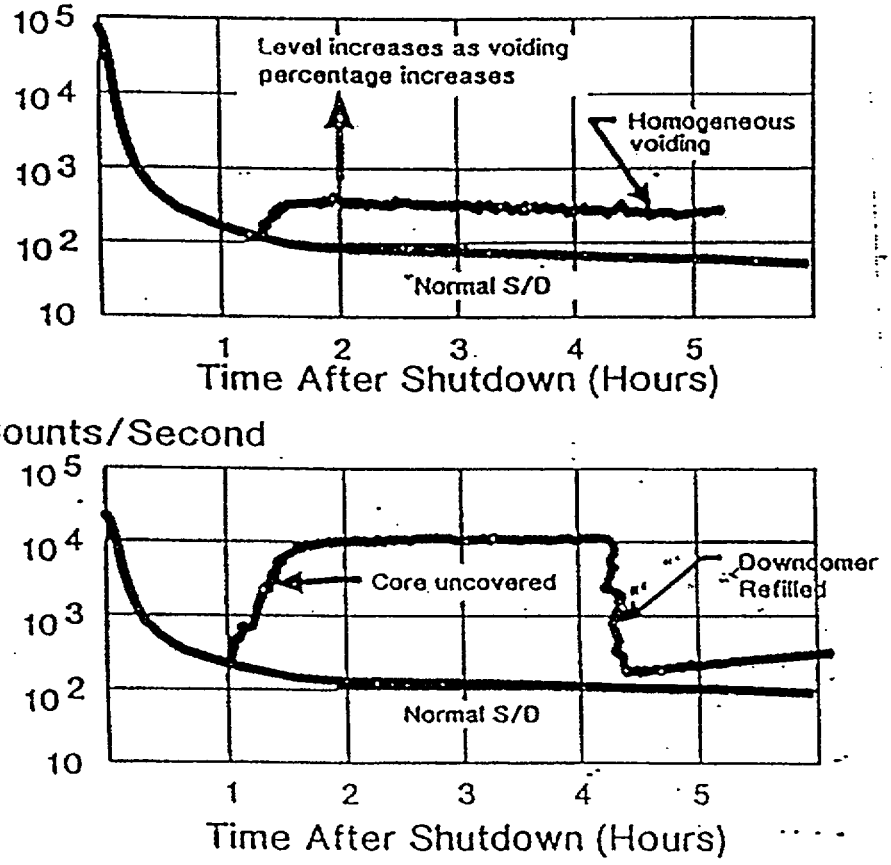
Thermocouple			Temperature Readings		[1] Corrections	
Number	Core Location	OAC Point ID	Incore Inst Panel Deg F.	OAC Deg F.	Corrected Panel Deg F	Corrected OAC Deg F
T35	B-5	A1275				
T36	B-9	A0113				
T37	B-11	A1281				
T39	D-3	A0131				
T40	D-7	A0137				
T41	D-11	A0143				
T42	D-13	A1287				
T43	F-1	A0155				
T46	F-13	A0173				
T48	H-1	A0185				
T49	H-3	A0191				
T50	H-7	A0197				
T51	H-11	A1311				
T52	H-15	A0209				
T53	K-1	A1317				
T54	K-5	A1323				
T55	K-9	A0227				
T56	K-13	A1326				
T57	K-15	A1335				
T58	M-3	A0245				
T60	M-11	A1341				
T61	M-13	A1348				
T63	P-5	A1360				
T64	P-7	A0281				
T65	P-9	A1366				

NOTE: 1. To make correction add the reference junction difference to the recorded readings.

Prepared By/Date:_____

_____ 4.2.1 Obtain a plot of the Source Range Detectors (OAC points A1177 and A1206) response since they were energized following the shutdown. If available attach a hard copy of the plots to this Enclosure.

NOTE:	For a detailed description of the possible accident response of the excore instrumentation see "Accident Response of Instrumentation" of the Westinghouse Advanced Station Nuclear Engineer Training.



_____ 4.2.2 Using the plots above as a guide, examine the Source Range plots from Step 4.2.1 to determine if the core experienced voiding and for how long.

Core Experienced Voiding ☐ YES ☐ NO

IF Core Experienced Voiding, What was the Duration of the Voiding: _____hrs

Enclosure 4.3

Movable Incore Detector Response

NOTE:	**IF** the Keithley picoammeter is not able to be set up or if it is not desirable to set it up, the ENA system can still be used to estimate the 3-D location of core damage by attempted insertion of the detectors.

_____ 4.3.1 Set up ENA System for special low level measurement per OP/0/A/6150/007 (Incore Instrumentation), Section 8.

_____ 4.3.2 Select thimble locations trying to put at least one detector in each quadrant and one near the center of the core.

_____ 4.3.3 Record the selected thimble locations in the table in step 4.3.8 (Page 2 of this enclosure).

NOTE:	Operate the ENA System in MANUAL LOW SPEED.

_____ 4.3.4 Insert the detectors to top of core.

_____ 4.3.5 **IF** any detector will not insert to top of core complete the following steps in this section.

1. Record actual distance reading the detector(s) were able to be inserted in the table in step 4.3.8 (Page 2 of this enclosure).

2. Record expected Top Of Core, from Enclosure 4.3, Table 2.7 of the applicable Units Data Book in the table in step 4.3.8 (Page 2 of this enclosure).

3. Calculate the difference in inches on 4.3.8 (Page 2 of this enclosure) and record on the Core Layout in step 4.3.9 (Page 3 of this enclosure).

NOTE:	For detailed description of the possible accident response of the incore instrumentation see "Accident Response of Instrumentation" of the Westinghouse Advanced Station Nuclear Engineer Training.

_____ 4.3.6 Obtain a flux trace and attach to this enclosure.

_____ 4.3.7 Compare flux traces with the anticipated traces on Pages 4 and 5 of this enclosure ("Anticipated Low Level Trace With Partial core Voiding" and "Anticipated Low Level Trace With Significant Core Damage) and identify/record indications of core voiding and/or core damage.

_____ 4.3.8 Complete the Following Table

PASS		Drive A	Drive B	Drive C	Drive D	Drive E	Drive F
1	Thimble Location						
	Expected T.O.C.						
	Actual Reading						
	Difference (inches)						
2	Thimble Location						
	Expected T.O.C.						
	Actual Reading						
	Difference (inches)						
3	Thimble Location						
	Expected T.O.C.						
	Actual Reading						
	Difference (inches)						
4	Thimble Location						
	Expected T.O.C						
	Actual Reading						
	Difference (inches)						
5	Thimble Location						
	Expected T.O.C						
	Actual Reading						
	Difference (inches)						
6	Thimble Location						
	Expected T.O.C.						
	Actual Reading						
	Difference (inches)						

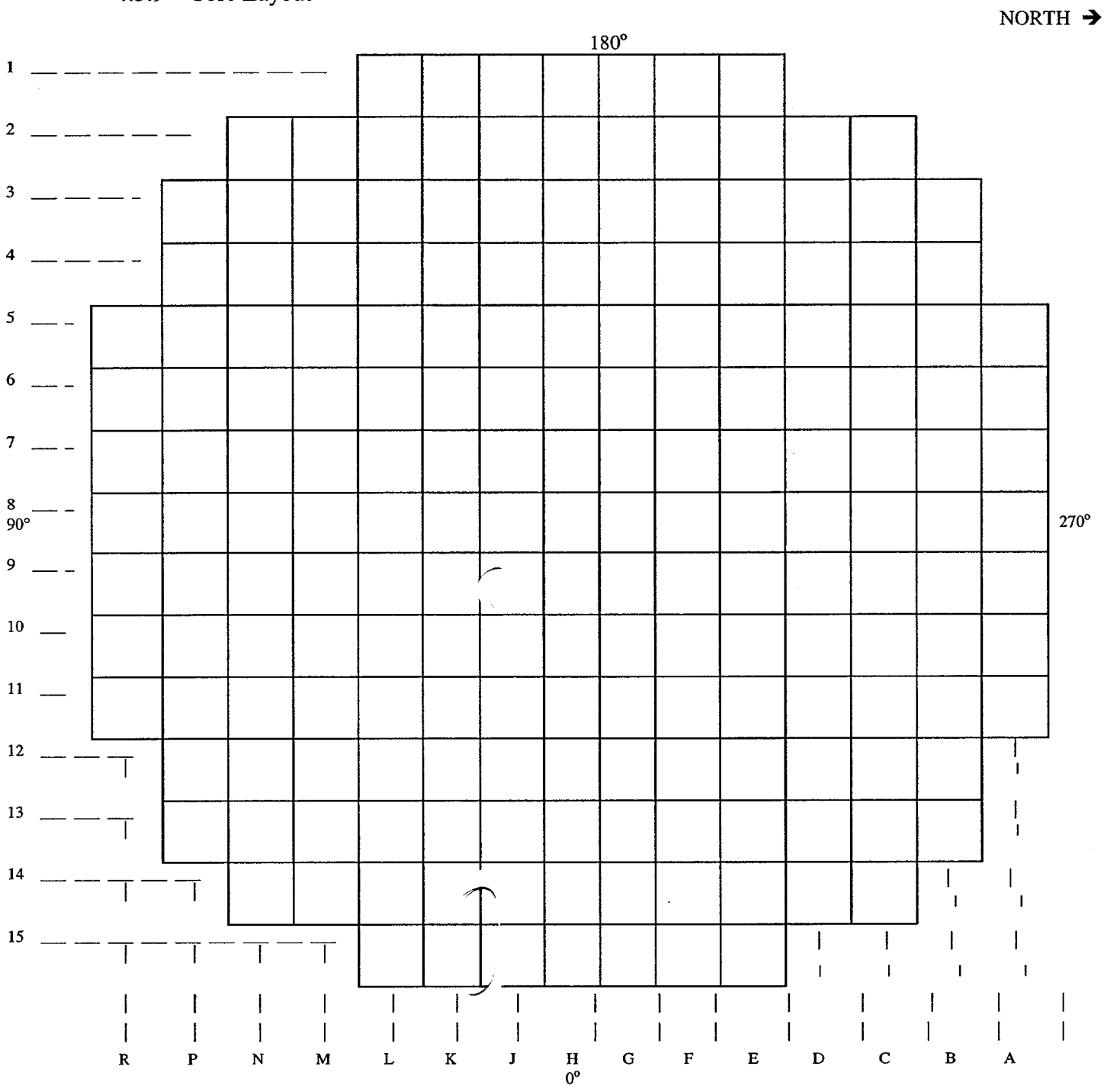
NOTE:

1) T.O.C. = Top of Core nixie tube readout

2) Difference = (Expected T.O.C.) - (Actual Reading)
This difference represents where core damage impairs incore movement relative to the T.O.C.

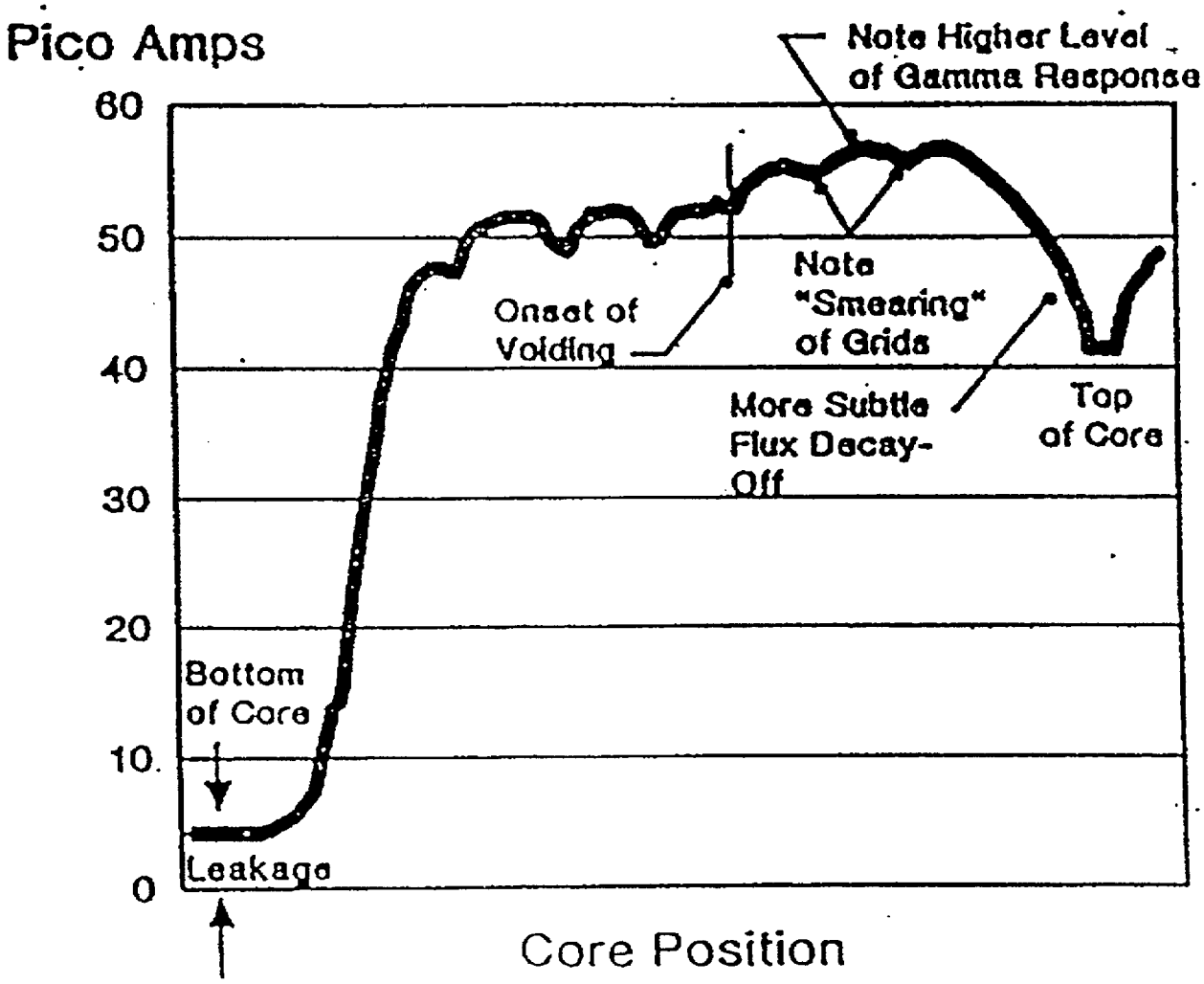
3) Expected T.O.C. can be found in Enclosure 4.3, Table 2.7 of the applicable Unit's Data Book.

4.3.9 Core Layout

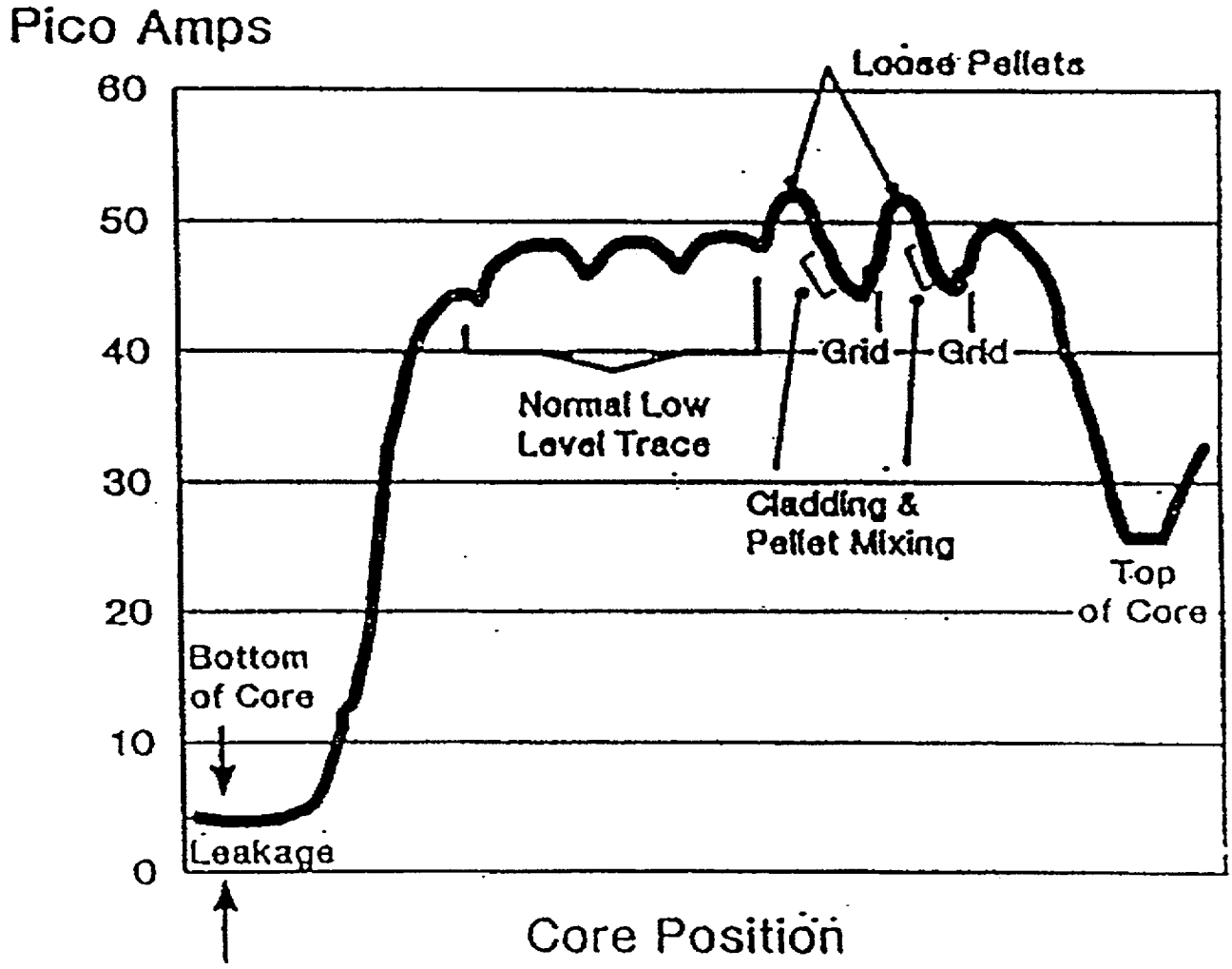


NOTE:	Enter difference between expected and measured detector insertion value in appropriate thimble location.

ANTICIPATED LOW LEVEL TRACE
WITH PARTIAL CORE VOIDING
(Gamma Response)



ANTICIPATED LOW LEVEL TRACE
WITH SIGNIFICANT CORE DAMAGE
(Gamma Response)



RESULTS

☐ Possible Fuel Rod **Clad Damage** (Step 3.2)	☐ Possible Fuel **OT Damage** (Step 3.3)
_____% Damage Based on CRM's (Step 3.2.4)	_____% Damage Based on CRM's (Step 3.3.4)
_____% Damage Based on CET's (Step 3.2.5)	_____% Damage Based on CET's (Step 3.3.5)

Comments from Step 3.2.6 or Step 3.3.6/Additional Comments:

The points below make up the NUC01 group on the OAC.

This can be displayed by typing "GD NUC01" to bring up the group display.

OAC Point Ids	Description
P0828	5 Highest CET's (Deg F)
P1457	Burnup, EFPD
A1047	Containment Pressure A (psig)
A0665	Containment Pressure B (psig)
A1041	Containment Sump Level A (Ft)
A0671	Containment Sump Level B (Ft)
A0829	EMF 51A (R/hr)
A0835	EMF 51B (R/hr)
A0848	Hydrogen Concentration A (%)
A0854	Hydrogen Concentration B (%)
P0755	Lower Cont Weighted Avg Temperature (Deg F)
P1461	NC Loops Temp AVG-AVG From OK Loops, (Deg F)
P0829	NC Pressure, Best Estimate (psig)
P1470	Pressurizer Level, Best Estimate (%)
A1306	RVLIS Low Range A (%)
A1330	RVLIS Low Range B (%)
A1177	Source Range Level Channel 1 (cps)
A1206	Source Range Level Channel 2 (cps)
A1204	Upper Containment Ambient Air Temp. A
A1210	Upper Containment Ambient Air Temp. B